From: "Vanderkamp, David W." <dwvande@nppd.com>
To: "'mct@nrc.gov'" <mct@nrc.gov>
Date: 8/22/02 3:19AM
Subject: FW: Updated version of draft response to RAIs (MSIV leakage path)

<<NRC-RAIs-SA RESPONSES-RLY1.wpd>>
Here is the updated version of the draft response. Sorry, it is not a redline/strikeout. Just toss the first version I sent. The attached file will be the one used during the teleconference. The engineer that is to be the lead during the teleconference will be unavailable next week. It would be good if we could schedule the teleconference for this week.

David Van Der Kamp
Cooper Nuclear Station
Licensing
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Attachment 7

Mail Envelope Properties (3D649077.A8C : 17 : 60044)

Subject: FW: Updated version of draft response to RAIs (MSIV leakage path)
Creation Date: 8/21/02 2:53PM
From: "Vanderkamp, David W." <dwvande@nppd.com>

Created By: dwvande@nppd.com

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MCT (Mohan Thadani)

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Files	**Size**	**Date & Time**
MESSAGE	468	08/21/02 02:53PM
NRC-RAIs-SA RESPONSES-RLY1.wpd		1370177
Mime.822	1876891	

Options

Expiration Date:	None
Priority:	Standard
Reply Requested:	No
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Concealed Subject:	No
Security:	Standard

ATTACHMENT 1
NPPD RESPONSES TO USNRC
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1. In your submittal (Reference 1), you indicated that the 2.0xSSE [safe shutdown earthquake] ground response spectrum (GRS) envelopes the floor response spectra (FRS) at elevation 932'-6" in both the Control Building (CB) and Reactor Building. However, Figure 4.5 shows that the FRS at elevation 932'-6" in the CB is higher than the 2.0xSSE GRS. Explain the discrepancy. Also, provide a figure, which confirms that the 2.0xSSE GRS envelopes the FRS at elevation 903'-6" in the CB.

Response: Figure 4.5 from Reference 1 (EE 01-147, Rev. 0, Attachment 9.2, Page 5) is duplicated below as Figure 1. The FRS for CB 903'-6" has been added to this figure, which shows that 2.0xSSE GRS completely envelops the FRS for CB 903'-6".

As stated above, 2.0xSSE GRS does not completely envelop the FRS for CB 932'-6". While 2.0xSSE GRS does not completely envelop the CB 932'-6" FRS, 2.0xSSE GRS was judged to be an adequate estimate of the FRS at TB 932' because:

1. Figure 2 shows that 2.0xSSE GRS is a conservative envelop of the RB 931'-6" FRS, and,
2. the points at which the CB 932'-6" FRS is higher than 2.0xSSE GRS are relatively small, and they occur in the low frequency regions, below about 2 Hz.

Note that the Control Building and Reactor Building FRS were calculated using the IPEEE RLE, which, as shown in Figure 3, has substantially more low frequency content than either the CNS SSE or the Regulatory Guide 1.60 GRS. Even so, these conservatively calculated FRS are enveloped by the 2.0xSSE GRS in all but the low frequency region which is the subject of this RAI.

The above discussion is moot at this point, because, as stated in Reference 4, 2.0xSSE GRS is no longer being used as an estimate of the TB FRS. Instead, specific TB FRS have been calculated following the guidance in NUREG-0800 (Standard Review Plan) Sections 3.7.1 and 3.7.2.

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2. You indicated that the methodology described in NUREG/CR-6240 (Reference 2) was used to determine the seismic capacity of welded and non-welded (e.g , threaded pipe) steel piping. Indicate whether NRC has reviewed and accepted the methodology as an acceptable approach to determine the seismic capacity of the steel piping.

Response: The direct reference to NUREG/CR-6240 will be removed from EE 01-147, and the direct tie to BWROG Report NEDC-31858P, Appendix D and the associated USNRC Safety Evaluation, will be reinforced within EE 01-147. In addition, the technical basis for the knockdown factors applied to threaded and other non welded fittings will be supplied. The following is a general summary of the application of this knockdown factor, and is provided for information only at this time:

> The criteria (capacity and allowable spans) established in Appendix D of NEDC-31858P, Appendix D and subsequent BWROG submittals are based primarily on welded steel pipe having yield stresses on the order of 25,000 psi to 35,000 psi. The experience data contains examples of non-welded fittings (threaded, cast iron, etc.) and non-ductile materials (cast iron) which have not performed as well as welded steel pipe in strong motion earthquakes. To address these situations in the experienced based evaluation of the piping in the CNS MSIV Leakage Pathway, a one-third (1/3) reduction in the allowable unsupported spans was applied to piping systems containing threaded fittings. Systems containing cast iron components are classified as outliers and require a more detailed review and evaluation. It should be noted that the systems containing these types of components are a very small percentage of the CNS MSIV Leakage Pathway.

This approach is essentially the same as that accepted in Paragraph 5 of Section 4.5 of the NRC's Safety Evaluation for the Monticello MSIV Leakage Pathway Submittal.

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3. You indicated in Reference 1 that the seismic demand for outlier resolution will be 2 times the GRS in the horizontal direction and 2/3 the GRS in the vertical direction for all piping systems. The 2/3 the GRS in the vertical direction is based on an assumption that there is no amplification of the vertical seismic input ground motion by the Turbine Building (TB). Justify the TB is perfectly rigid in the vertical direction.

Response: Reference 1 stated that, 2/3 the GRS will be used as the vertical seismic demand because:

1. this was consistent with the vertical seismic demand stated in the CNS USAR for all Class I structures (including the Control and Reactor Buildings), and
2. the Turbine Building is a substantial reinforced concrete shear wall structure from the foundation mat (El. 877') to the operating deck (El. 932'), and it was reasonable to assume that it too would not amplify the vertical ground motion.

Note that the above discussion is moot, because, as stated in Reference 4, multiples of the GRS are no longer being used as an estimate of the TB FRS. Instead, specific TB FRS have been calculated following the guidance in NUREG-0800 (Standard Review Plan) Sections 3.7.1 and 3.7.2. In addition, the analyses utilized in the generation of these specific TB FRS have shown the first vertical structural mode of the building to be in excess of 20 Hz, which is above the frequency regions in which significant amplification of the seismic ground motion occurs. This is concluded to provide adequate justification for the previous use of 2/3 of the GRS in the vertical direction.

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4. You indicated in Reference 1 that the anchor bolt capacities of Appendix C of the Seismic Qualification Utility Group-Generic Implementation Procedure (SQUG-GIP) (Reference 3) will be used for the pipe support evaluations. However, if anchor bolts exist that are not given in the SQUG-GIP, then the manufacturer's capacities will be used with a factor of safety 3.0. Discuss your justification for not using the manufacturer's recommended factor of safety.

Response: The capacities for concrete expansion anchors (CEAs) in Appendix C of SQUG-GIP are based on a factor of safety of 3.0 (see EPRI NP-5228-SL Rev.1, Vol. 1, Table 2.16). A safety factor of 3.0 would have been used for other CEAs in order to maintain consistency with SQUG-GIP. However, no "non-GIP" CEAs were encountered in evaluating the pipe supports at CNS. Most of the CEAs are Phillips "Self-Drilling" expansion anchors. A few Phillips "wedge" anchors were also encountered. Both of these CEAs are covered by SQUG-GIP.

Some of the pipe supports (particularly the larger Main Steam Line supports) are anchored using the original cast-in-place Richmond Inserts, or a combination of Richmond Inserts and concrete expansion anchors. SQUG-GIP addresses certain cast-in-place anchors, but not any which are deemed similar to Richmond Inserts. The Richmond Inserts were evaluated using the capacity values specified in CNS procedures, which were determined through the use of the manufacturer's recommended factor of safety.

Thus, there were no actual instances where anchorage evaluations were performed which did not utilize the manufacturer's recommended factor of safety, or the SQUG-GIP approved values.

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5. You used Equation 5.9 in Reference 1 for determining the adequacy of the anchor bolt capacity. Discuss how Equation 5.9 is more conservative than the bilinear formulation given in the SQUG-GIP (Reference 3).

Response: The two interaction curves are shown below in Figure 4. When the shear ratio (V/V_{all}) is less than about 0.4, the elliptical curve is more conservative. When the shear ratio is greater than about 0.4, the SQUG-GIP curve is more conservative.

SQUG-GIP primarily addresses the evaluation of equipment and equipment anchorage. In conducting the equipment anchorage evaluation for this program, the SQUG-GIP interaction curve was applied. For pipe supports, the test data summarized in Reference 7.26 (EPRI Report TR-101968, Tier 2, Volume 3, "Guidelines and Criteria for Nuclear Piping and Support Evaluation and Design", Volume 3: Anchor Bolt Capacity and Installation Tolerances) of EE 01-147 recommends that the elliptical relationship better represents test results for pipe supports. Thus, while the elliptical curve is not more conservative than the bilinear formulation in all possible combinations of shear/tension interaction, the elliptical curve is considered to be more appropriate for pipe support evaluations.

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6. Equations 5.1a through 5.3 in Reference 1 are similar to the equations contained in the ASME Boiler and Pressure Vessel Code, Section III, Division 1 for Class 3 piping systems. If the ASME type equations are used for a piping evaluation, then the appropriate *i* factor (stress intensification factor) from the version of ASME Code where those equations appear should be used in the evaluation.

Response: See the response to question 7.

7. You stated that the basis for the establishment of Equation 5.3 in Reference 1 is that" ... S_A for carbon steel pipe is approximately 1.5 S_y which is approximately 5/8 S_y, The majority of the piping is A-106B GR B CS with S = 15,000 psi and S_y = 36,000 psi so, 2.5 S_A (2.5 × 1.5 × 15,000) = 56,250 psi and, therefore, 2.5 S_A is approximately 1.6 S_y, The applied stresses are secondary; limiting the range of applied stress to less than 2 S_y ensures that elastic shakedown will occur, no significant membrane stress rupture will occur, and the accumulated cyclic damage will be elastic. Therefore, given the limited number of cycles of strong motion in a Design Basis SSE (10 to 20 cycles) and that elastic cycling below the 2.0 S_y will occur, a fatigue failure due to the SAMs from one SSE would not occur. Therefore, the 1.6 S_y secondary stress range limit used is significantly less than the upper bound limit of 2 S_y and with this limit no fatigue failures due to one SSE event would be anticipated."

However, the NRC staff has a different view on Equation 5.3. Equation 5.3 specifies the use of ½ the range of SSE anchor moments. This justification implies that the range of anchor motions is held to less than 2 S_y, Your statement is not accurate unless Equation 5.3 considers the full range of SSE. Provide your discussion with respect to the staff's view.

Response: The reference to ASME BPVC Section III, Division 1 was made as a convenient source to provide a technical basis for the 2.4 S_h limit being used. The reference to ASME Section III will be deleted from this discussion in EE 01-147. The revised basis for Equation 5.1 and 5.2 is provided as follows:

Equation 5.1 is the deadweight allowable stress equation per the B31.1 Power Piping Code. In equation 5.2, S is the allowable material stress per the B31.1 Power Piping Code, which is the lesser of 5/8 S_y (2/3 S_y in the later code editions) or S_u/4. The majority of the piping under review is A-106B carbon steel pipe, which has S = 15,000 psi, S_y = 35,000 psi, and S_u = 60,000 psi. Therefore equation 5.2 limits the pressure + deadweight + SSE seismic inertial stress combination to less than 1.03 S_y, which ensures elastic behavior. Further, it ensures the validity of the linear elastic analysis techniques used in the static and dynamic analyses that were conducted.

(Items (a) and (b) under the paragraph in EE 01-147, Rev. 0 which begins with the sentence, "The basis for the establishment of equation 5.2 is as follows:" will be replaced with the above discussion.)

In addition to equation 5.3, for all dynamic and static analyses, the following equation was checked:

$i[(2 \times M_{bsam}) / z] \leq 2.5S_A$; where i, M_{bsam}, z, S_A are as defined in EE 01-147, Rev. 0

This was done to address the case where the amplitude (½ range) of SAM stress is larger than the thermal stress. This equation (in conjunction with equation 5.3) ensures that the worse secondary stress range is limited to approximately 1.6 S_y, which is less than 2.0S_y and will ensure that elastic shakedown will occur. Therefore, given the limited number of strong motion SSE cycles (10 cycles to 20 cycles), there will be no fatigue failures due to one SSE event. This equation and the above discussion will be added to EE 01-147.
In regards to Equation 5.3, the reference to ASME Section III will be removed. The basis for the equation will then be provided as follows:

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In equation 5.3, S_A for carbon steel pipe is approximately 1.5 S, which is approximately 22,500 psi, and, therefore, 2.5 S_A is approximately 1.6 S_y. These stresses are secondary in nature, and, therefore, if they are limited to less than 2.0 S_y (per ASME criteria document, "Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Section III and Section VIII, Division 2"), this will ensure that elastic shakedown will occur, no significant membrane stress rupture will occur, and accumulated cyclic damage will be elastic. The 1.6 S_y limit used here is significantly less than the upper bound 2.0 S_y limit, and is an acceptable secondary stress limit.

[Items (a) and (b) under section 4.5.4.2.1, Piping, in EE 01-147, Rev. 0 which begins with the sentence, "The basis for the establishment of equation 5.3 is as follows:" will be replaced with the above discussion. In addition, a new reference, ASME criteria document, "Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Section III and Section VIII, Division 2" will be added to the Reference Section of EE 01-147.]

This is essentially the same stress criteria the USNRC previously accepted for the evaluation of the MSIV leakage pathway for the Monticello Plant.

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8. You stated in Reference 1 that "Recent criteria and studies including Regulatory Guideline 1.61 [Damping Values for Seismic Design of Nuclear Power Plants], the ASME Boiler and Pressure Vessel Code Section III, Division 1, Appendix N, and NUREG/CR-0098 specify levels of damping for the SSE analysis of piping systems. In all the aforementioned documents, the basis of the determination of damping values is primarily the stress level in the component, not the basis or methodology used for response spectrum generation. That is, once a response spectrum is selected, the specified damping is based on the response of the structure under analysis in terms of fabrication methods and member stress levels. Newmark and Hall in NUREG/CR-0098, specify damping values of 2% to 3% for piping stressed to no more than ½ S_y and 5% to 7% for piping stressed to approximately the yield point. The ASME Boiler *and Pressure* Vessel Code, Section 111, Division 1, Appendix N, currently specifies 5% damping for the evaluation of the piping systems at both the Level B and Level D conditions. The Level D condition corresponds to the SSE event under evaluation here."

The NRC staff does not agree with your statement. The basis for staff acceptance of 5 percent damping is the conservatism in the spectra generation. This position has. been previously stated in the NRC endorsement of Code Case N-411 in Regulatory Guide 1.84 [Design and Fabrication Code Case Acceptability-ASME Section III Division 1).

Response: As stated in Reference 4, the median-centered estimate of the floor response spectra (2.0×SSE GRS) is no longer being utilized. Instead, specific floor response spectra have been calculated following the guidance in NUREG-0800 (Standard Review Plan) Sections 3.7.1 and 3.7.2, using a Regulatory Guide 1.60 spectral shape as the input ground response spectrum. Therefore, the new spectra being used are conservative. This, in addition to the reasons previously cited in EE 01-147, Rev. 0, justifies the use of 5% damping. However, it should also be noted that in Section 4.3.1 (a) of the USNRC's Safety Evaluation for the Monticello MSIV Leakage pathway, the staff accepted the use of 5% damping in conjunction with an approximate median-centered spectra for the seismic ruggedness evaluation of piping systems in the Monticello MSIV Leakage Pathway.

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9. You indicated in Reference 1 that an approach called the "collapsed beam" approach is used for localized evaluation of piping systems. The NRC staff is not aware of the "collapsed beam" approach and did not endorse the approach previously. Justify the reasons why the "collapsed beam" approach is equivalent to or more conservative than the analysis methods discussed in Sections 3.9.1 and 3.9.2 of the NRC Standard Review Plan.

Response: The terminology "Collapsed Load Method" simply means the use of classical beam theory with conservatively established spans and end conditions to conduct a static stress analysis of local portions of piping using manual methods. This method is the same as the equivalent static load method permitted in Section 3.9.2 of NUREG-0800. The terminology "Collapsed Beam Method" will be removed from EE 01-147, Rev. 0 and replaced with the terminology "Equivalent Static Load Method". This method is essentially the same method which was previously accepted by the USNRC in Paragraph 3 of Section 4.6.1 of the USNRC's Safety Evaluation for the Monticello MSIV Leakage Pathway Submittal.

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10. During the teleconference held on May 8, 2002, the licensee indicated that the piping support components at Cooper Nuclear Station (CNS) are designed in accordance with the requirements in MSS-SP-58, "Pipe Hangers and Supports - Materials, Design, and Manufacture." In Reference 1, the licensee indicated that the capacities of the piping support components for the Level D load case should not exceed 2.0 times the capacities specified in MSS-SP-58 based on the ASME Boiler *and Pressure Vessel* Code Case N-500-1. The NRC staff requests response to the following:

10(a) The ASME *Boiler and Pressure Vessel Code* Case N-500-1 specified other requirements (e.g., materials, quality assurance program, etc.) in order to use 2.0 times the capacities specified in MSS-SP-58 for the Level D load case Indicate whether the piping support components at CNS meet the pertinent requirements of the ASME Code that would permit an increase in the load capacity by a factor of 2.0 times at the load Level D.

Response: ASME Code Case N-500-1 was used as a convenient source to provide an acceptable technical basis for the use of Service Level D capacities for standard support components of 2 times the MSS-SP-58 capacities. The technical basis of this Code Case is independent of the material traceability requirements. The reference to ASME Code Case N-500-1 will be deleted, and the technical basis for the use of the factor of 2.0 will be directly provided in EE 01-147, Rev. 0. The basis is as follows (and is based on the MSS-SP-58 (1967 edition) which would be consistent with the B31.1.0-1967 edition that is the code of record for Cooper Nuclear Station):

Section 4 of MSS-SP-58, (1967 edition) requires that in establishing pipe support component capacities, the maximum stress shall not exceed the allowable stresses in Table 2 of MSS-SP-58. Furthermore, for materials not listed in Table 2, the maximum allowable stress shall be $S_u/5$. The materials in Table 2 include several grades of cast and malleable iron. These materials are not germane to the discussion here as they are screened out during the walkdown as "non-ductile" support components and require special consideration and evaluation.

A review of the allowable stress (for temperatures up to 650 °F) for steel bars, plates, bolts, straps and castings in Table 2, shows that the minimum ratio of the yield stress to the allowable stress is 2.0. Therefore, using the rated loads for the MSS-SP-58 components maintains the working stresses in the component to no higher than ½ S_y. This is for the minimum ratio of the yield stress to the Table 2 allowable stress. The average ratio for all materials is 0.4 S_y. Therefore, multiplying the MSS-SP-58 rated capacities by a factor of 2 results in support components that are an average of 0.8 S_y, and, in the limiting the case, support components that are at the material yield stress. Therefore, multiplying the MSS-SP-58 rated capacities by a factor of 2 ensures essentially elastic behavior for the SSE event and ensures no significant deformation of the supports. Further, it ensures the validity of the linear elastic analysis techniques used in the static and dynamic analyses that were conducted.

The approach of keeping the support component allowable stresses at or slightly below the material yield stresses is consistent with the criteria and capacities the USNRC accepted in Section 4.6.2 of the Safety Evaluation for the Monticello MSIV Leakage Pathway Submittal.

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10(b) In Reference 4, the licensee indicated that CNS Updated Safety Analysis Report specifies the use of 0.9 S_y as the stress limit for the piping support components for the Level D load case. This limit exceeds 2.0 times the capacities specified in MSS-SP-58. Provide justification for suggesting to use an even higher limit than those permitted in the ASME Boiler and Pressure Vessel Code Case N-500-1. Also, indicate whether NRC had reviewed and accepted your use of 0.9 S_y as a stress limit for the piping support components at CNS for the Level D load case.

Response: In Reference 4 (NLS2002073), NPPD documented its intention to pursue a course of action to establish higher capacities for some component standard supports based on the greater of either:

1. 0.9 S_y, <u>or</u>
2. 2.0 × the MSS-SP-58 specified allowable stresses for the faulted condition (with SSE loading).

NPPD's original submittal (NLS2002014 and EE 01-147, Rev. 0) specified a criteria of only 2.0 × the MSS-SP-58 specified allowable stresses for the faulted condition.

As discussed in the response to 10 a), the <u>rated loads</u> for the MSS-SP-58 components maintain the working stresses in the component to no higher than ½ S_y (i.e., $S_{allowable} = 0.5\ S_y$). This value corresponds to the material(s) with the minimum ratio of the material yield stress to the Table 2 allowable stress (i.e , S_y / $S_{allowable} = 2$) . The average ratio the material yield stress to the Table 2 allowable stress for all materials is approximately 2.5 which equates to an average allowable stress of 0.4 S_y. Therefore, multiplying the MSS-SP-58 rated capacities by a factor of 2 results in support component members that are on average at about 0.8 S_y and in the limiting case support component members are at the material yield stress (1.0 S_y). For example, for grade 55, A663-82 carbon steel (for rods & bars), 2 times the listed allowable stress is 0.996 S_y [(2 × 13.7) / 27.5]. **Therefore, the use of an allowable stress capacity of 0.9 S_y for the detailed evaluation of component support items is consistent with the 0.8 S_y to 1.0 S_y capacity achieved with the use of 2.0 times the MSS-SP-58 rated loads.**

Neither the CNS Updated Safety Analysis Report (USAR) nor any other CNS design basis document, previously reviewed by the NRC, explicitly addresses the usage of 0.9 S_y as the stress limit for **<u>catalog component standard supports</u>** for the Level D load case; however, they (CNS USAR and/or NEDC 92-017) do explicitly address the usage of 0.9 S_y for Class I pipe support structural members, Class I building structural steel, structural steel rebar used in CNS Class I concrete structures, Class I conduit and cable tray supports, etc. when subjected to the SSE load cases.

NPPD's original licensing basis, as approved by the NRC, for using 0.9 S_y for the specified SSCs was that no "loss of function" would occur for the SSE load cases (reference FSAR Amendment 9, question 12.9). The criteria for "no loss of function" is that <u>stresses remain in the elastic range</u>. This criteria is also based on the low probability of the SSE event.

Some of the applicable CNS catalog component standard supports are attached to and supported directly by a structural steel pipe support structure that is governed by an allowable stress limit of 0.9 S_y. Therefore, it is logical to conclude that the catalog component standard support could also use this same criteria.

Therefore, NPPD believes that the use of 0.9 S_y is consistent with previously approved criteria at CNS and provides sufficient margin against failure of the applicable pipe supports. Therefore, NPPD requests explicit NRC approval of the aforementioned course of action **<u>to be used only for the pipe supports within the scope of the MSIV leakage pathway seismic qualification project</u>**. This approach meets

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the intent of the NRC's "Safety Evaluation of GE Topical Report, NEDC-31858P, Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems," September 1993" to "provide reasonable assurance that the main steam line...will maintain structural integrity and operability during and following an SSE." This criteria also ensures "no loss of function" as previously defined.

The approach of keeping the support component allowable stresses at or slightly below the material yield stresses is consistent with the criteria and capacities the USNRC accepted in Section 4.6 2 of the Safety Evaluation for the Monticello MSIV Leakage Pathway Submittal.

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11. In Reference 4, the licensee indicated that a numerical technique (i.e., finite element analysis) will be used to establish the capacities of the pipe support components. Discuss your rationale for concluding that a finite element analysis, which relies on approximation of the geometry, can be considered to provide a more realistic estimate of the load carrying capacity of the analyzed component than the actual testing performed by the vendor for such component.

Response: MSS-SP-58 (1967 edition) requires that support components be load rated such that the allowable material stress remains less than the allowable stresses given in Section 4 of MSS-SP-58. There is no explicit or implied requirement in MSS-SP-58 that these load ratings be established by test. They can be, and in many cases are, established by calculation or analysis. The use of FEA analysis is simply a more refined method of analysis which can be used to establish a support component load rating consistent with MSS-SP-58.

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12. In Reference 4, the licensee indicated that it will use the concrete anchor bolt capacities used in IE Bulletin 79-02 [Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts] with a factor of safety of 4. However, IE Bulletin 79-02 requires a factor of safety larger than 4 for certain types of anchor bolts. Provide your technical justification for using only the factor of safety of 4.

Response: For the evaluation of existing supports, the actual intention on the use of concrete anchor bolt capacities, is to use the highest capacity from either the SQUG-GIP, Appendix C data <u>or</u> from the CNS response to IEB 79-02 data, with the IEB 79-02 data reconciled for 5,000 psi concrete and a factor of safety of 4. NPPD has demonstrated that the use of 5,000 psi for the concrete compressive strength is appropriate for Turbine Building anchorages in EE 02-016, Evaluation of Compression Strength of Concrete for the Turbine Building. For most of the applicable concrete anchors, the SQUG-GIP capacities, which are based on a factor of safety of 3 and 4,000 psi concrete, are higher than the corresponding IEB 79-02 capacities when using a factor of safety of 4. However, for some anchor sizes, a slightly higher capacity than the SQUG-GIP values is achieved from the revised IEB 79-02 data. Since the SQUG-GIP uses a factor of safety of 3, and typically yields higher capacities than the corresponding IEB 79-02 data using a factor of safety of 4, it was concluded that the use of the factor of safety of 4 was an acceptable approach for the applicable concrete anchors.

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References:

1. Letter, Nebraska Public Power District to U.S. NRC, "License Condition 2.C.(6) Seismic Evaluation, Cooper Nuclear Station, NRC Docket No 50-298, DPR-46," dated February 26, 2002.

2. NUREG/CR-6240, Stevenson & Associates, "Application of Bounding Spectra to Seismic Design of Piping Based on the Performance of Above Ground Piping in Power Plants Subject to Strong Motion Earthquakes," February, 1995.

3. Seismic Qualification Utility Group, "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Power Plant Equipment," Revision 2, corrected February 14, 1992.

4. Letter, Nebraska Public Power District to U.S. NRC, "Supplemental Information Related to License Condition 2.C.(6) Seismic Evaluation, Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46," dated June 9, 2002.

5. Regulatory Guide 1.60

6. NUREG-0800, Standard Review Plan

7. NEDC-31858P, Revision 2, General Electric, "BWROG Report for Increasing MSIV Leakage Ratio Limits and Elimination of Leakage Control Systems," September 1993, (principally Appendix D thereof).

8. USNRC Safety Evaluation of GE Topical Report, NEDC-31858P, Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems," September 1993

9. CNS Updated Safety Analysis Report, Section XII and Appendix C

10. CNS FSAR Amendment 9, Question 12.9

11. EPRI/SQUG, "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Revision 2, February 1992.

12. EPRI Report NP-5228-SL, Rev.1, Vol. 1, Table 2.16

13. EPRI Report TR-101968, Tier 2, Volume 3, "Guidelines and Criteria for Nuclear Piping and Support Evaluation and Design", Volume 3: Anchor Bolt Capacity and Installation Tolerances

14. ASME criteria document, "Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Section III and Section VIII, Division 2"

15. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 102 to Facility Operating License No. DPR-22 Northern States Power Company Monticello Nuclear Generating Plant Docket No. 50-263

16. MSS-SP-58, "Pipe Hangers and Supports - Materials, Design, and Manufacture", 1967 Edition

17. ASME Code Case N-500-1

References (Cont.):

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18. NPPD Calculation NEDC 92-017, Rev. 2, Class IS Large Bore Piping and Pipe Support Analysis Guideline

19. CNS "Pipe Support Base Plate Designs Using Expansion Anchor Bolts Response to NRC Bulletin 79-02" Dated 10-7-80

20. EE 02-016, Rev. 0, Evaluation of Compression Strength of Concrete for the Turbine Building

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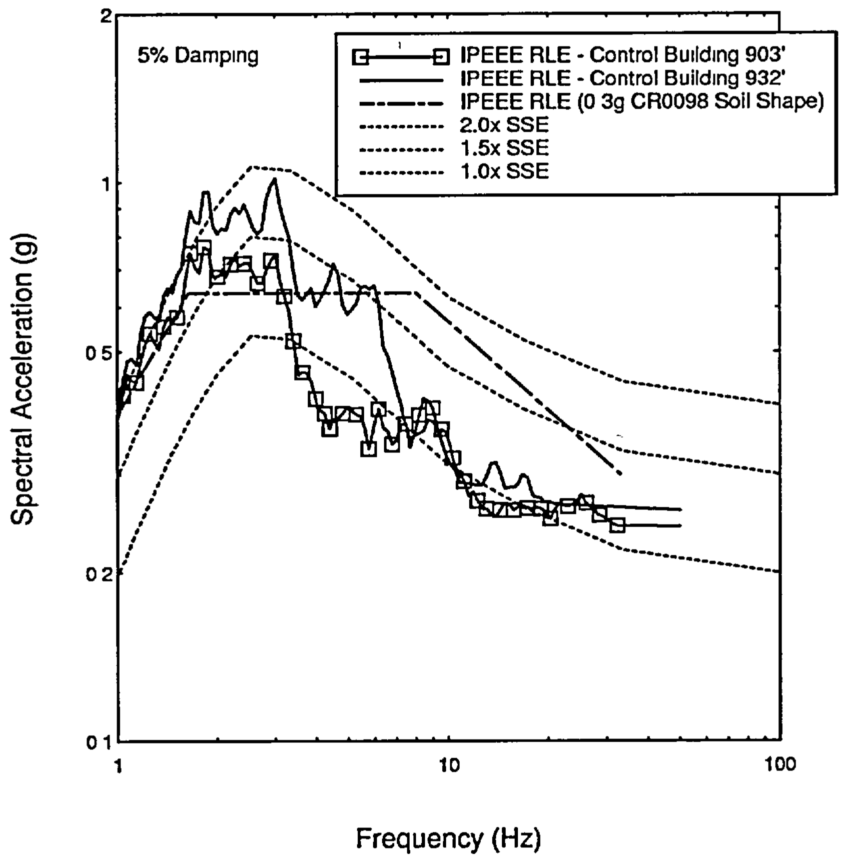

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Figure 1. IPEEE RLE Floor Response Spectra for the Control Building

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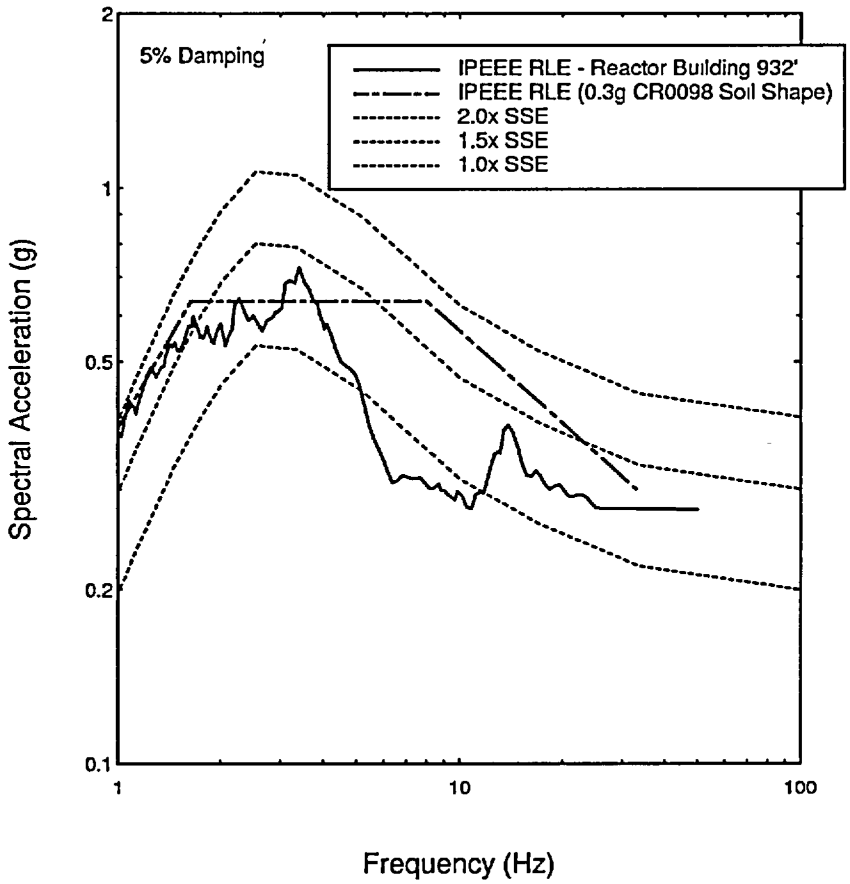

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Figure 2. IPEEE RLE Floor Response Spectra for the Reactor Building

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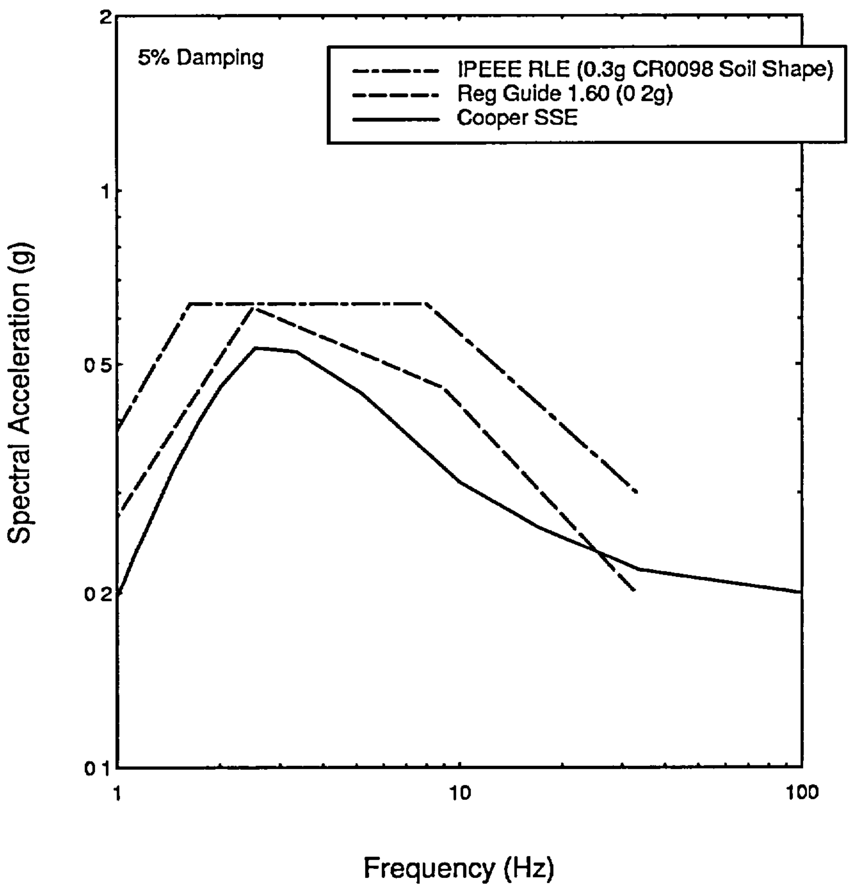

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Figure 3. Comparison of Ground Response Spectra

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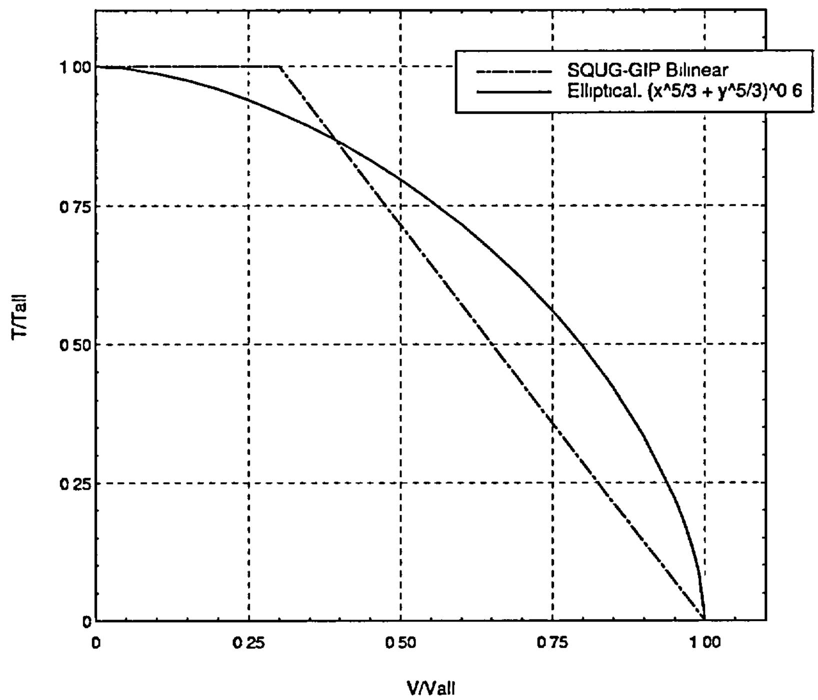

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Figure 4. Comparison of Shear (V) / Tension (T) Interaction Curves